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Docket No. 50-298

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing & Quality Assurance
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

In response to your requests dated September 16, November 2, and December 23, 1977, the Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS).

The amendment involves changes in the CNS Technical Specifications.to add to the list of primary containment isolation valves four valves which had been inadvertently omitted, and to correct a typographical error on page 8 of the Technical Specifications. The amendment also changes the Administrative Controls section of the Technical Specifications to add qualification requirements for the Chemistry and Health Physics Supervisor, to alter the method of identifying the Safety Review and Audit Board members, to revise the Nebraska Public Power District management organization chart, to delete the respiratory protection program requirements and to modify certain reporting requirements.

To meet our requirements, certain changes to the Technical Specifications which you proposed were necessary. These changes have been discussed with and agreed to by your staff.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. to DPR-46
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

OFFICE		ORB#2:DOR	ORB#2:DOR	OELD	ORB#2:DOR
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DATE		nm 1/24/78	1/23/78	1/27/78	1/31/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

cc w/enclosures:
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Columbus, Nebraska 68601

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Phoenix, Arizona 85004

Cooper Nuclear Station
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Station Superintendent
P. O. Box 98
Brownville, Nebraska 68321

Auburn Public Library
118 - 15th Street
Auburn, Nebraska 68305

Director
Department of Environmental Control - w/copy of NPPD filings dtd. 9/16/77, 11/2/77 & 12/23/77
Executive Building, 2nd Floor
Lincoln, Nebraska 68509

Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nemaha County Courthouse
Auburn, Nebraska 68305

Chief, Energy Systems Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VII
ATTN: EIS COORDINATOR
1735 Baltimore Avenue
Kansas City, Missouri 64108

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Nebraska Public Power District dated September 16, November 2 and December 23, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance JANUARY 31 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Technical Specifications contained in Appendix A of the above indicated license with the attached pages bearing the same numbers (except as otherwise indicated). Changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
iii of Table of Contents	iii of Table of Contents
8	8
174	174
219	219
-	219a (new)
223	223
226	226
227	227
228	228
229	229
230	230
231	231
232	232
233	233
234	234
235	-
236	236
237	-
238	-
239	-
240	-
241	-
242	-
243	-

Note: Some of the above pages are reissued to renumber pages as a result of deletion of a number of pages regarding respiratory protection equipment.

TABLE OF CONTENTS (cont'd.)

2.1.A (Cont'd)

d. APRM Rod Block Trip Setting

The RPRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\%$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2381 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2 million lb/hr.)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the setting shall be modified as follows:

$S_{RB} \leq (0.66\ W + 42\%)\ \frac{A}{MTPF}$

Where:

A = 2.64 for 7x7 feel
= 2.44 for 8x8 fuel

MTPF = the value of the existing maximum total peaking factor

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

$\geq$+12.5 in. on vessel level instruments.

TABLE 3.7.4 (page 2)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-36	CRD-11CV and CRD-12CV, CRD Exhaust Water	Air
X-39A	RHR-MO-26A and RHR-MO-31A, Drywell Spray Header Supply	Air
X-39B	RHR-MO-26B and RHR-MO-31B, Drywell Spray Header Supply	Air
X-41	RRV-740AV and RRV-741AV, Reactor Water Sample Line	Air
X-42	SLC-12CV and SLC-13CV, Standby Liquid Control	Air
X-205	PC-233MV and PC-237AV, Purge and Vent Supply to Torus	Air
X-205	PC-13CV and PC-243AV, Torus Vacuum Relief	Air
X-205	PC-14CV and PC-244AV, Torus Vacuum Relief	Air
X-205	MV-1303 AND MV-1304, ACAD Supply to Torus	Air
X-210A	RCIC-MO-27 and RCIC-13CV, RCIC Minimum Flow Line	Air
X-210A	RHR-MO-21A, RHR to Torus	Air
X-210A	RHR-MO-16A, RHR-10CV, and RHR-12CV, RHR Minimum Flow Line	Air
X-210B	RHR-MO-21B, RHR to Torus	Air
X-210B	HPCI-17CV and HPCI-MO-25, HPCI Minimum Flow Line	Air
X-210B	RHR-MO-16B, RHR-11CV, and RHR-13CV, RHR Minimum Flow Line	Air
X-210A and 211A	RHR-MO-34A, RHR-MO-38A, and RHR-MO-39A, RHR to Torus	Air
X-210B and 211B	RHR-MO-34B, RHR-MO-38B, and RHR-MO-39B, RHR to Torus	Air
X-212	RCIC-15CV and RCIC-37, RCIC Turbine Exhaust	Air
X-214	HPCI-15CV and HPCI-44, HPCI Turbine Exhaust	Air
X-214	HPCI-AO-70 and HPCI-AO-71, HPCI Turbine Exhaust Drain	Air
X-214	RHR-MO-166A and RHR-MO-167A RHR Heat Exch. Vent	Air
X-214	RHR-MO-166B and RHR-MO-167B RHR Heat Exch. Vent	Air
X-220	PC-230MV and PC-245AV, Purge and Vent Exhaust from Torus	Air
X-221	RCIC-12CV and RCIC-42, RCIC Vacuum Line	Air
X-222	HPCI-50 and HPCI-16CV, HPCI Turbine Drain	Air

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 The Station Superintendent shall have the over-all full-time onsite responsibility for the safe operation of the Cooper Nuclear Station. During periods when the Station Superintendent is unavailable, he may delegate his responsibility to the Assistant to Station Superintendent or, in his absence, to one of the Department Supervisors.

6.1.2 The portion of the Nebraska Public Power District management which relates to the operation of this station is shown in Figure 6.1.1.

6.1.3 The organization for conduct of operation of the station is shown in Fig. 6.1.2. The shift complement at the station shall at all times meet the following requirements. Note: Higher grade licensed operators may take the place of lower grade licensed or unlicensed operators.

A. A licensed senior reactor operator (SRO) shall be present at the station at all times when there is any fuel in the reactor.

B. A licensed reactor operator shall be in the control room at all times when there is any fuel in the reactor.

C. Two licensed reactor operators shall be in the control room during all startup, shutdown and other periods involving significant planned control rod manipulations. A licensed SRO shall either be in the Control Room or immediately available to the Control Room during such periods.

D. A licensed senior reactor operator (SRO) with no other concurrent duties shall be directly in charge of any refueling operation, or alteration of the reactor core.

A licensed reactor operator (RO) with no other concurrent duties shall be directly in charge of operations involving the handling of irradiated fuel other than refueling or reactor core alteration operations.

E. An individual who has been trained and qualified in health physics techniques shall be on site at all times that fuel is on site.

F. Minimum crew size during reactor operation shall consist of three licensed reactor operators (one of whom shall be licensed SRO) and two unlicensed operators. Minimum crew size during reactor cold shutdown conditions shall consist of two licensed reactor operators (one of whom shall be licensed SRO) and one unlicensed operator.

In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury a qualified replacement shall be designated to report on-site within two hours.

6.1.4 The minimum qualifications, training, replacement training, and retraining of plant personnel at the time of fuel loading or appointment to the active position shall meet the requirements as described in the ANSI-N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants". The Assistant to Station Superintendent qualifications shall comply with Section 4.2 of ANSI-N18.1-1971. The Chemistry and Health Physics Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, Sept. 1975;

6.1.4 (Cont'd) personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

6.2 (cont'd)

1. Membership

   a. Assistant General Manager (chairman)

   b. Director of Licensing and Quality Assurance (alternate Chairman)

   c. Director of Power Projects

   d. Director of Power Supply

   e. Director of Environmental Affairs

   f. Consultants (as required)

   The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized.

   Alternate members shall be appointed in writing by the Board Chairman to serve on a temporary basis; however, no more than two alternates shall serve on the Board at any one time.

2. Meeting frequency: Semiannually, and as required on call of the Chairman.

3. Quorum: Chairman or Vice Chairman, plus three members including alternates. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Responsibilities: The following subjects shall be reported to and reviewed by the NPPD Safety Review and Audit Board.

   a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

   b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

6.3 Station Operating Procedures

6.3.1 Station personnel shall be provided detailed written procedures to be used for operation and maintenance of system components and systems that could have an effect on nuclear safety.

6.3.2 Written integrated and system procedures and instructions including applicable check off lists shall be provided and adhered to for the following:

A. Normal startup, operation, shutdown and fuel handling operations of the station including all systems and components involving nuclear safety.

B. Actions to be taken to correct specific and foreseen potential or actual malfunctions of safety related systems or components including responses to alarms, primary system leaks and abnormal reactivity changes.

C. Emergency conditions involving possible or actual releases of radioactive materials.

D. Implementing procedures of the Security Plan and the Emergency Plan.

6.3.3 The following maintenance and test procedures will be provided to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

A. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

B. Routine testing or standby and redundant equipment.

C. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

D. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

E. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

6.3.4 Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR 20.

6.4 Actions to be taken in the Event of Occurrences Specified in Section 6.7.2.A

6.4.1 Occurrences, as specified in Section 6.7.2.A., shall be promptly reported to the Station Superintendent, Director of Power Supply and the Chairman of the NPPD Safety Review and Audit Board and shall be promptly reviewed by the Station Operations Review Committee. This committee shall prepare a separate report. This report shall include an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the occurrence. Copies of all such reports shall be submitted to the Power Supply Department and the NPPD Safety Review and Audit Board Chairman for review and approval of any recommendations.

6.4.2 All occurrences as specified in Section 6.7.2.A. shall be reported to the General Manager on a periodic basis.

6.5 Action to be Taken if a Safety Limit is Exceeded

6.5.1 If a safety limit is exceeded, reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC. An immediate report shall be made to the Director of Power Supply, the General Manager and to the chairman of the NPPD Safety Review and Audit Board. A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Station Operations Review Committee. This report shall be submitted to the Director of Power Supply and the NPPD Safety Review and Audit Board. Appropriate analyses or reports will be submitted to the NRC. Notification of such occurrences will be made to the NRC by the Station Superintendent within 24 hours as specified in Specification 6.7.

6.6 Station Operating Records

6.6.1 Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years unless a longer period is required by applicable regulations.

A. Records of normal station operation, including power levels and periods of operation at each power level.

B. Records of periodic checks, inspection and/or calibrations performed to verify that Surveillance Requirements are being met.

C. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

D. Records of occurrences and safety limit violations as specified in 6.4 and 6.5.

E. Record of changes to plant procedures.

F. Records of special tests and experiments.

G. Records of wind speed and direction.

6.6.2 Records and logs relating to the following items shall be kept for the life of the plant.

A. Records of changes made to the station as described in the Safety Analysis Report and amendments and reflected in updated, corrected and as-built drawings and records.

B. Records of new and spent fuel inventory and assembly histories.

C. Records of station radiation and contamination surveys.

D. Records of off-site environmental monitoring surveys.

E. Records of radiation exposure for all station personnel, including all contractors and visitors to the station in accordance with 10 CFR 20.

F. Records of radioactivity in liquid and gaseous wastes released to the environment.

G. Design Fatigue Usage Evaluation

1. Monitoring, recording, and evaluation will be met for various portions of the reactor coolant pressure boundary (RCPB) for which detailed fatigue

usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The locations to be monitored shall be:

a. The feedwater nozzles

b. The shell at or near the waterline

c. The flange studs

2. Monitoring, Recording, Evaluating, and Reporting

a. Operational transients that occur during plant operations will, at least semi-annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report Code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specification quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicted number of design condition events by twenty-five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

H. Records of individual plant staff members showing qualifications, training and retraining.

6.6.3 Records and logs relating to the following items shall be kept for two years.

A. The test results, in units of microcuries, for leak tests of sources performed pursuant to Specification 3.8.A.

B. Records of annual physical inventories verifying accountability of the sources on record,

---

1. See paragraph N-415.2, ASME Section III, 1965 Edition.

2. The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition.)

6.7 Station Reporting Requirements

6.7.1 Routine Reports

A. In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate NRC Regional Office of Inspection and Enforcement unless otherwise noted.

B. Start up Report

1. A summary report of plant startup and power escalation testing shall be submitted following:

   a. Receipt of an operating license.

   b. Amendment to the license involving a planned increase in power level.

   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.

   d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

   The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

2. Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

C. Annual Reports

Routine reports covering the subjects noted in 6.7.1.C.1, 6.7.1.C.2, 6.7.1.C.3 and 6.7.1.C.4 for the previous calendar year shall be submitted prior to March 1 of each year.

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 6.6.2G, Design Fatigue Usage, a listing of the number of events identified in 6.6.2.G.2.b will be tabulated and compared to the design or allowed quantity of comparable or more severe events. In those cases where recalculation of fatigue usage is required per 6.6.2.G.2.c and the calculated usage exceeds two times the design usage limit of the Code, the report will define the inservice inspections that will be performed on that portion of the RCPB to monitor for crack initiation.

4. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. Monthly Operating Report
Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report.

6.7.2. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

1/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

A. <u>Prompt Notification With Written Follow-up</u>. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office, or his designate no later than the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 6.7.2.A.5, 6.7.2.A.6 or 6.7.2.B.1 below.

2. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.7.2.B.2 below.

3. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

4. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of any unplanned criticality.

5. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

6. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

   Note: For items 6.7.2.A.5 and 6.7.2.A.6 reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.7.2.B.2 and 6.7.2.B.3 below.

7. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

8. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

9. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

B. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

1. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

2. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.7.2.B.1 and 6.7.2.B.2 need not be reported except where test results themselves reveal a degraded mode as described above.

3. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

4. Abnormal degradation of systems other than those specified in item 6.7.2.A.3 above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.7.3. Unique Reporting Requirements

Reports shall be submitted to the Director, Nuclear Reactor Regulation, USNRC, Washington, D. C. 20555, as follows:

A. Reports on the following area shall be submitted as noted:

	Area	Reference	Submittal Date
1.	Secondary Containment Leak Rate Testing (1)	4.7.C.1	90 days after completion of each test.

NOTE: (1) Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

Amendment No. 17, 41

NPPD MANAGEMENT
ORGANIZATION CHART

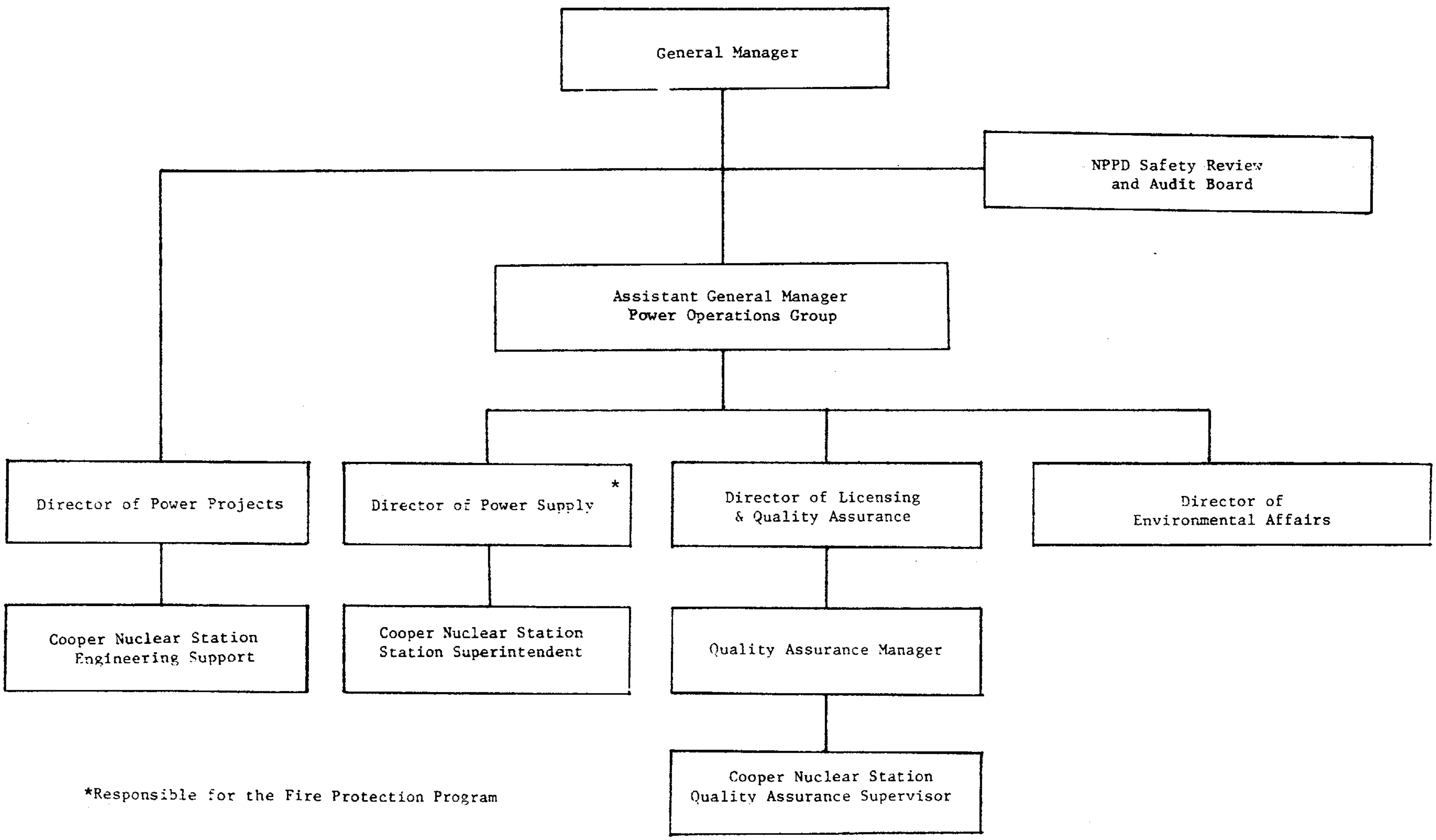
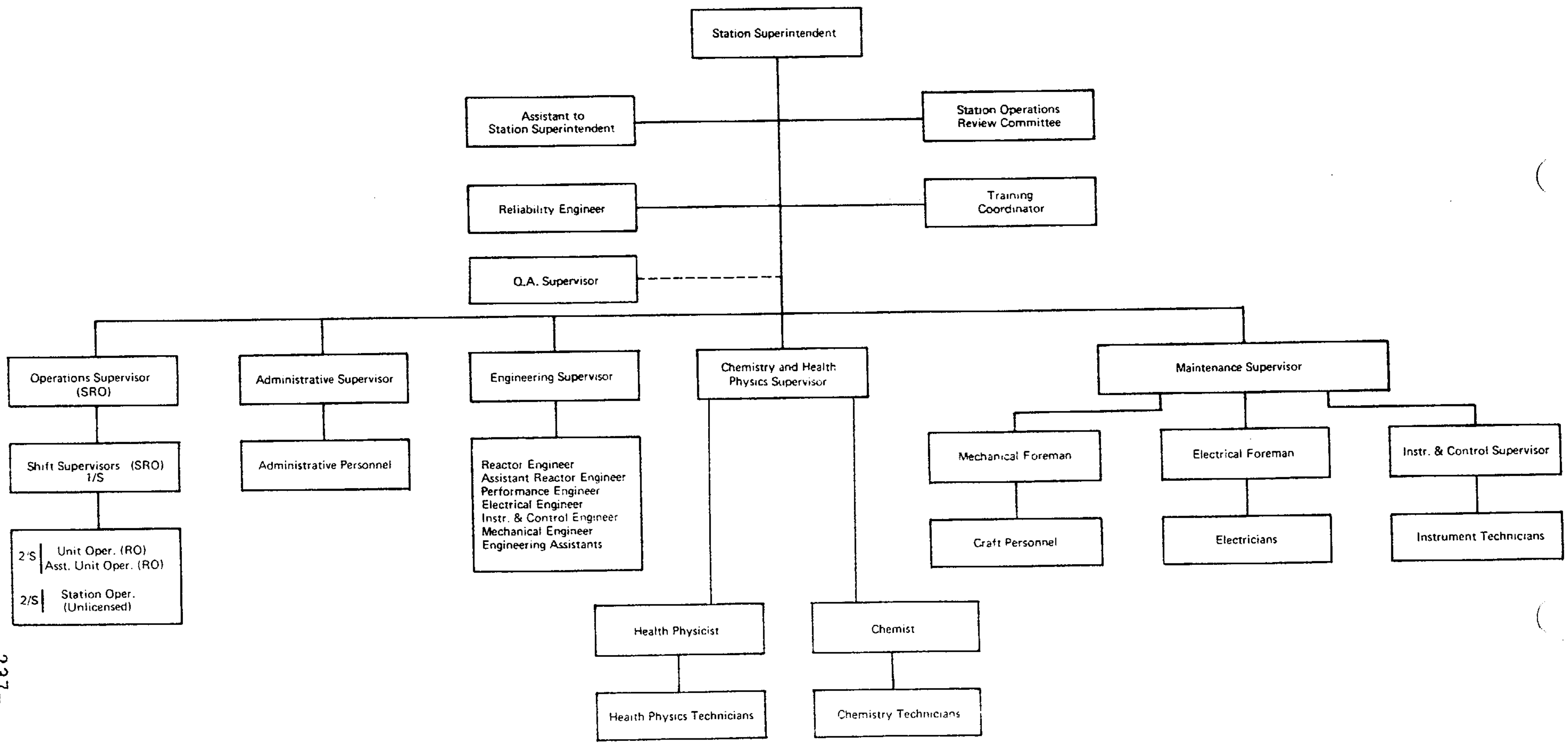


Figure 6.1.1
NPPD Management
Organization Chart

Amendment No. 17, 41

CNS ORGANIZATION CHART



1/S one/shift
2/S two/shift

RO AEC Reactor Operators License
SRO - AEC Senior Reactor Operators License

Figure 6.1.2
Cooper Nuclear Station
Organizational Chart

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

This Safety Evaluation supports several changes to the Appendix A Technical Specifications for the Cooper Nuclear Station (CNS). These changes involve the addition of four valves to the list of primary containment testable isolation valves and changes to the Administrative Controls section of the Technical Specifications to (1) add qualification requirements for the Chemistry and Health Physics Supervisor, (2) alter the method of identifying the Safety Review and Audit Board (SRAB) members, (3) revise the Nebraska Public Power District (NPPD) management organization chart, (4) delete the respiratory protection program section from the Technical Specifications, and (5) modify certain reporting requirements. Each of the changes is addressed in a separate section of the evaluation. Modifications to certain of the changes proposed by NPPD were required to meet NRC requirements. These modifications were discussed with and accepted by NPPD. In addition to the above changes, a minor change has been incorporated into the equation for control rod block setpoint on page 8 of the Technical Specifications (section 2.1.A.d). This change corrects a typographical error by replacing the "greater than or equal to" sign in the equation with the correct "less than or equal to" sign.

PRIMARY CONTAINMENT ISOLATION VALVES

Discussion/Evaluation

By letter dated September 16, 1977, the licensee proposed the addition of four valves to Table 3.7.4 which is a list of primary containment testable isolation valves. The four valves were identified as containment isolation valves during a detailed review of the Residual Heat Removal (RHR) system at CNS. However, these valves had not been previously identified as containment isolation valves in Table 3.7.4.

We have evaluated the proposed addition to Table 3.7.4 and found it acceptable because it would provide a more complete and, therefore, more correct listing of the testable primary containment isolation valves in the CNS Technical Specifications.

CHEMISTRY AND HEALTH PHYSICS SUPERVISOR QUALIFICATIONS

Discussion

In a letter dated March 15, 1977, we informed the licensee of our position concerning the qualifications of the individual who performs the function of Radiation Protection Manager (RPM). Our position, as stated in Regulatory Guide 1.8, September 1975, is that if the RPM is reassigned or the incumbent is replaced, the new RPM should have qualifications equivalent to those stated in the Regulatory Guide. To implement this position, we requested the licensee to propose a Technical Specification to be included in the Administrative Controls section. The specification would state that the RPM, or equivalent position title (Chemistry and Health Physics Supervisor at CNS), shall meet or exceed the qualifications of Regulatory Guide 1.8.

By letter dated September 16, 1977, the licensee proposed to include the requisite statement in the CNS Technical Specifications; however, the licensee's proposal included a qualifying statement that "personnel qualifications equivalent to those stated in the Regulatory Guide may be proposed in selected cases".

Evaluation

We have evaluated the proposed specification and determined that it is acceptable provided the qualification statement is modified to state "personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases". Since the implementation section of Regulatory Guide 1.8 permits the applicant to propose alternative qualifications for complying with specific portions of the Commission's regulations, we find the proposed Technical Specification, as modified, to be acceptable.

SAFETY REVIEW AND AUDIT BOARD MEMBERSHIP

Discussion

In the September 16, 1977 Technical Specification proposal, the licensee requested to delete the Safety Review and Audit Board (SRAB) membership specification which identified board members by functional title. In place of this specification, a statement which authorized the Assistant General Manager, Power Group, to appoint SRAB members, as necessary, was proposed. The licensee stated that this would provide that the appropriate expertise was maintained despite internal organizational changes.

Evaluation

We have determined that the licensee's proposed Technical Specification change does not meet our requirements for continuity of board membership. To meet our requirements, the licensee agreed to modify his proposal to incorporate the membership requirements specified in the standard specifications which we use in developing Technical Specifications for new facilities. Although this results in a reduction from 8 to 5 in the number of SRAB member position titles identified, we have concluded that the proposed change would result in no reduction in the review and audit capability of the SRAB because the Board quorum requirements remain unchanged and Specification 6.2.1.B.1 requires that:

> "The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved."

Thus, use of the standard specification does provide some additional flexibility in appointing SRAB members despite internal organization changes and also meets our requirements for membership continuity. Therefore, we find the proposed Technical Specification, as modified, to be acceptable.

NPPD ORGANIZATION CHANGES

Discussion/Evaluation

By letter dated December 23, 1977, NPPD submitted a revised organization chart for inclusion in the CNS Technical Specifications. The organizational changes were:

1. The title of "Director of Generation Engineering and Construction, to whom the CNS Station Superintendent reports, has been changed to "Director of Power Projects" and this position now reports directly to the office of the General Manager.

2. The Director of Power Supply has been designated as the responsible corporate individual for the Fire Protection Program at CNS.

We have evaluated these proposed changes and found them acceptable because the functions of the Director of Power Projects would remain essentially unchanged and the effectiveness of the CNS management chain should be improved by the direct link between the Director of Power Projects and the NPPD General Manager. Also, the designation of an individual in corporate management as responsible for the fire protection program at CNS fulfills a requirement deriving from the ongoing NRC fire protection review of nuclear plants.

RESPIRATORY PROTECTION PROGRAM

Discussion/Evaluation

On November 29, 1976, The Commission published in the Federal Register an amended Section 20.103 of 10 CFR 20, which became effective on December 29, 1976. One effect of this revision is that in order to receive credit for limiting the inhalation of airborne radioactive material, respiratory protective equipment must be used as stipulated in Regulatory Guide 8.15. Another requirement of the amended regulation is that licensees authorized to make allowance for use of respiratory protective equipment prior to December 29, 1976, must bring the use of their respiratory protective equipment into conformance with Regulatory Guide 8.15 by December 29, 1977. Because the respiratory protective program described in the Administrative Controls section of the CNS Technical Specifications differs from that in Regulatory Guide 8.15, we informed the licensee, by letter dated August 12, 1977, that the respiratory protection program portion of the Technical Specifications would be deleted in a future license amendment.

We have concluded that the elimination of the respiratory protection section of the CNS Technical Specifications is acceptable because, with the change in the regulations (10 CFR 20.103), there is no longer a need for specific authorization for use of respiratory protection equipment in the Technical Specifications.

REPORTING REQUIREMENTS

Discussion/Evaluation

In a letter dated September 19, 1977, we requested that the licensee (1) delete from the Technical Specifications the requirement for an Annual Operating Report provided that certain information presently reported in the Annual Operating Report continues to be reported and (2) modify the content of the Monthly Operating Report. By letter dated November 2, 1977, the licensee proposed the above changes to the reporting requirement Technical Specifications.

We have evaluated the proposed changes and concluded that they are acceptable because:

(1) the Annual Operating Report in the present Technical Specifications can be deleted and sufficient information will still be available to meet the NRC reporting objectives; and

(2) the modifications to the Monthly Operating Report will provide more timely information for our evaluation.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 31, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 41 to Facility Operating License No. DPR-46, issued to the Nebraska Public Power District (the licensee), which revised the Technical Specifications for operation of the Cooper Nuclear Station (the facility) located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

The amendment involves changes in the facility Technical Specifications to add four valves to the list or primary containment isolation valves, to add qualification requirements for the Chemistry and Health Physics Supervisor, to alter the method of identifying Safety Review and Audit Board members, to revise the licensee's management organization chart, to delete the respiratory protection program requirements, and to modify certain facility reporting requirements.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commisssion's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated September 16, November 2, and December 23, 1977, (2) Amendment No. 41 to Licensee No. DPR-46, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commisssion's Public Document Room 1717 H Street, N. W., Washington, D. C. and at the Auburn Public Library, 118 - 15th Street, Auburn, Nebraska 68305. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 31st day of January, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors